NON-PUBLIC?: N
ACCESSION #: 9304190317
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Salem Generating Station Unit #2 PAGE: 1 OF 6

DOCKET NUMBER: 05000311

TITLE: Reactor Trip From 100% Power Due to 22 SGFP Trip on Low
Suction Pressure
EVENT DATE: 03/16/93 LER #: 93-005-00 REPORT DATE: 04/15/93

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR
SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: M.J. Pollack LER Coordinator TELEPHONE: (609) 339-2022

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: SJ COMPONENT: FCV MANUFACTURER: C635
X SB RV M120
X SD PS S382
X SD RLY M120

REPORTABLE NPRDS: Y
Y
Y
Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On 3/16/93, at 1106 hour , subsequent to a 22 Steam Generator Feedwater Pump (SGFP) trip on low suction pressure, a reactor trip occurred on 24 Steam Generator (SG) low-low level. Condensate Polisher (CP) vessel regeneration was in progress. Following the reactor trip, 24 SG level increased rapidly (due to the 24BF19 valve failing open) and feedwater isolation initiated upon 24 SG level reaching the high-high level setpoint. 24 SG was then isolated. The root cause of the reactor trip is "Management/Quality Assurance Deficiency". Less than timely actions to correct a leaking CPS valve packing and failure to mitigate increasing leakage resulted in failure of a differential pressure switch. This led to the 22 SGFP trip and subsequent Reactor Trip. The failed differential pressure switch was causing premature opening of 24CP2 (Feed and Condensate Inlet Valve). A pressure transient occurred resulting in a sudden drop of SGFP pump suction pressure sufficient to actuate one of the SGFP suction pressure trips. The differential pressure switch failed due to water intrusion. An aggressive leak reduction program is in place. The circumstances surrounding the events in this LER will be reviewed with applicable station personnel for increased sensitivity to mitigation of leaks. The planning effort in support of the 24CP14 valve repair will be reviewed. Additional corrective action will be taken based on the results of this review. The failed CP System differential pressure switch and its flex conduit were replaced. The 24CP14 packing leakage will be repaired.

END OF ABSTRACT

PLANT AND SYSTEM IDENTIFICATION:

Westinghouse - Pressurized Water Reactor

Energy Industry Identification System (EIIS) codes are identified in the text as {xx}

IDENTIFICATION OF OCCURRENCE:

Reactor Trip From 100% Power due to 22 SGFP trip on low suction pressure

Event Date: 3/16/93

Report Date: 4/15/93

This report was initiated by Incident Report No. 93-201.

CONDITIONS PRIOR TO OCCURRENCE:

Mode 1 Reactor Power 100% - Unit Load 1170 MWe

DESCRIPTION OF OCCURRENCE:

On March 16, 1993, at 1106 hours, subsequent to a 22 Steam Generator Feedwater Pump (SGFP) trip on low suction pressure, a reactor trip occurred on 24 Steam Generator (SG) low-low level.

Prior to the event, Condensate Polisher demineralizer vessel regeneration was in progress. At approximately 1104 hours, the Condensate Polisher System (CPS) operator heard a loud noise and received local CPS trouble alarms. At this time, the Control Room also received the CPS trouble alarms. These alarms cleared almost immediately and were followed by the 22 SGFP vibration OHA alarm and control console alarm indication of a 22 SGFP trip. CPS was immediately bypassed and a turbine runback initiated at 15% power per minute to 60% power, with control rods in automatic.

At 1106 hours, the reactor tripped on 24 SG low-low level. Following the reactor trip, 24 SG level increased rapidly and feedwater isolation initiated upon 24 SG level reaching the high-high level setpoint. 24 SG was isolated by closing the 24 SG Inlet Stop valve (24BF13) and the 24 SG Auxiliary Feedwater (AFW) level control valves (24AF11 and 24AF21).

At 1117 hours, manual Main Steamline Isolation was initiated, in accordance with Emergency Operating Procedure EOP-TRIP-2, due to excessive plant cooldown due to 24 SG being filled to 90%. Reactor Coolant System (RCS) {AB} temperature and pressure stabilized, although the 23 SG Atmospheric Relief Valve (23MS10) failed to respond

DESCRIPTION OF OCCURRENCE: (cont'd)

in automatic. The 23MS10 valve was successfully opened using manual control.

On March 16, 1993, at 1150 hours, the Nuclear Regulatory commission was notified of the reactor trip on 24 S/G low-low level in accordance with Code of Federal Regulations 10CFR 50.72 (b) (2) (ii)

APPARENT CAUSE OF OCCURRENCE:

The root cause of this event is "Management/Quality Assurance Deficiency". Less than timely actions to correct a leaking CPS valve packing and failure to mitigate increasing leakage resulted in failure of a differential pressure switch. This led to the 22 SGFP trip and subsequent Reactor Trip.

The CPS loud noise heard prior to the 22 SGFP trip was reproduced.

This was done by opening the Feed and Condensate Inlet Valve (24CP2) to repressurize the CP vessel (as had been done when the event first occurred). Investigation confirmed that a failed differential pressure switch (due to water intrusion) was causing premature opening of 24CP2 before pressure equalized between the demineralizer bed and condensate system. A pressure transient occurred resulting in a sudden drop of SGFP pump suction pressure sufficient to actuate one of the SGFP suction pressure trips. initial analysis indicates that a momentary drop of at least 127 psi occurred at the SGFP suction.

A possible contributing factor to the 22 SGFP trip was a reduction of SGFP suction pressure due to failure of the 2C Feedwater Heater Drain Tank level controller booster relay. This failure caused the 2C Feedwater Heater Drain Tank level control valve, 23HD9, to fail open. With tank level low, the 23HD15 valve (23 Heater Drain Pump Outlet Discharge Valve) closed resulting in reduced SGFP suction pressure thus making the SGFP more susceptible to a low suction pressure transient.

The differential pressure switch junction box door was found loose. The box enclosure contained water. This water failed the differential pressure switch. The water came from a 24CP14 valve packing leak. The leak was first identified in May 1992. Leakage occurs whenever CPS is in service. A collection device was placed below the leak redirecting the leak to a floor drain. An attempt to stop the leak, by tightening the packing nuts was unsuccessful. The valve was found to be obsolete consequently placing the work order on hold (awaiting engineering disposition). The leak, when first identified was approximately 5 gph. At the time of the Reactor Trip, the leak had progressed to approximately 5 gpm. It was recently identified that valve replacement parts are available (on special order). Upon receipt of parts the valve will be repaired.

In summary, management actions to ensure that periodic component

APPARENT CAUSE OF OCCURRENCE: (cont'd)

problems are addressed in a timely fashion, were not adequate to prevent this event.

ANALYSIS OF OCCURRENCE:

The Reactor Protection System (RPS) {JC} functioned as designed and the heat sink was maintained during this event. Also, since the RPS is designed for the thermal and hydraulic effects of four-hundred (400) full power reactor trips, this event involved no undue risk to the health or safety of the public. However, due to the RPS system actuation and the initiation of Main Steamline Isolation, the circumstances surrounding these events are reportable in accordance with Code of Federal Regulations 10CFR 50.7 3 (a) (2) (iv).

The low-low S/G level reactor trip prevents operation with S/G water level below the minimum required for adequate heat removal. The trip occurs on two out of three low-low level signals in any S/G. The setpoint ensures adequate S/G inventory, at the time of a reactor trip, to allow for possible Auxiliary Feedwater Pump starting delays; thus preventing S/G dryout and RCS thermal and hydraulic transients associated with a loss of the heat sink.

As identified in the Description of occurrence section, 24 SG level increased to the high-high level setpoint following the reactor trip. This was due to the 24BF19 main feedwater control valve failing to close. The 24BF19 valve is a Copes-Vulcan 14" air operated double seated globe valve (Model D-100-160-3). Investigation revealed two possible causes for the valve failure: 1) the four (4) tabs on the valve spring adjustment nut had broken off allowing the nut to rotate inside the yoke, possibly jamming itself inside the yoke and keeping the actuator hung up in the open position or 2) a piece of 1" schedule 80 pipe (21" long) found inside the valve cage below the plug could have held the valve open.

The 1" pipe (a fluted sampling pipe) was used to draw SG feedwater samples from the SG feedwater line prior to where the line separates into four (4) headers (to the 4 SGs). This pipe is cross sectionally installed within the SG feedwater line. investigation revealed that the 1" pipe had broken loose due to improper installation at original construction. The pipe was supported at one end instead of at both ends as required by design.

Following the start of the AFPs, the plant experienced an excessive cooldown. This was contributed to by the 24 SG level increase to 90% from failure of the 24BF19 valve. In accordance with Emergency Operating Procedure EOP-TRIP-2, a main steamline isolation (an ESF) was initiated stopping the cooldown. The plant was stabilized in mode 3 utilizing the MS10 atmospheric relief valves to maintain RCS temperature.

ANALYSIS OF OCCURRENCE: (cont'd)

After the manual main steamline isolation, 23 Turbine Driven AFW Pump was manually tripped, per the Emergency Operating Procedures, since it was not required for SG level control. When the SG low-low level signals cleared, the MS52 turbine trip valve was reset (i.e., opened) per the EOP, upon which, the turbine started to rotate. By design, once the start signals have been eliminated, the MS132 turbine start/stop valve solenoid should energize and the MS132 return to the closed position. Subsequent troubleshooting revealed that the 3A checking relay-auto (STOP) contact (in the auto start circuit) had a high resistance when the contact closed. This was sufficient to cause a voltage drop to prevent solenoid valve SV509 from energizing. This would cause the MS132 valve to remain open and the pump to subsequently start upon reopening the MS52 valve.

A similar reactor trip event occurred on September 4, 1990, (reference LER 311/90-036-00); although, its root cause was equipment failure. In that event, the 23HD15 valve failed closed due to its diaphragm rupturing. That failure coupled with a high SGFP suction pressure switch setpoint (due to drift) resulted in the loss of 21 SGFP. Operations had recovered from the loss of 21 SGFP; however, the 21BF19 valve failed closed causing increased feedflow to the other SGs leading to a 24 SG High-High Level Turbine Trip/Reactor Trip.

CORRECTIVE ACTION:

An aggressive leak reduction program is in place. The circumstances surrounding the events in this LER will be reviewed with applicable station personnel for increased sensitivity to mitigation of leaks.

The planning effort in support of the 24CP14 valve repair, since its leakage was first identified, will be reviewed. Additional corrective action will be taken based on the results of this review.

The failed CPS differential pressure switch and its flex conduit were replaced.

The remaining CPS differential pressure switches for the other 5 CPS vessels will be inspected.

The preventive maintenance requirements for the CPS differential pressure switches will be reviewed.

The heater drain system level control booster relay will be replaced.

The 24CP14 packing leak will be repaired prior to returning it to service.

Troubleshooting to determine why the 23MS10 valve failed to respond in automatic did not identify a specific component failure. MS10 control concern has been experienced during prior reactor trip events

CORRECTIVE ACTION: (cont'd)

(reference LER 311/92-014-00). This concern has been investigated by engineering. Modifications are planned for MS10 controls.

The 3A contact in the Turbine Driven AFW Pump autostart circuit will be repaired.

The Salem Unit 1 1" main feedwater sample pipe installation has been inspected and is installed correctly.

General Manager -
Salem Operations

MJP:pc

SORC Mtg. 93-033

ATTACHMENT 1 TO 9304190317 PAGE 1 OF 1

PSE&G

Public Service Electric and Gas Company P.O. Box 236 Hancocks Bridge, New Jersey 08038

Salem Generating Station

April 15, 1993

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Dear Sir:

SALEM GENERATING STATION
LICENSE NO. DPR-75
DOCKET NO. 50-311
UNIT NO. 2

LICENSEE EVENT REPORT 93-005-00

This Licensee Event Report is being submitted pursuant to the requirements of the Code of Federal Regulations 10CFR 50.73 (a) (2) (iv) This report is required to be issued within thirty (30) days of event discovery.

Sincerely yours,

C. A. Vondra
General Manager -
Salem Operations

MJP:pc

Distribution

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